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REQUEST FOR COMMENTS ON THE DRAFT REGULATORY GUIDE DG-1096

To Whom It May Concern:

Virginia Electric and Power Company (Dominion) appreciates the opportunity to provide comments on the draft Regulatory Guide DG-1096, "Transient and Accident Analysis Methods", as requested in the Federal Register, Vol. 65, No. 240 on December 13, 2000, page 77934.

We have reviewed the draft Regulatory Guide (DG) and submit the following comments for your consideration.

Dominion believes this DG and the associated references represent a valuable resource for code developers in standardizing and documenting the state of the art in safety analysis code development and application. However, we do have several concerns. The discussion which follows, and many of our concerns, center on the definition of an "evaluation model" and the implied scope of NRC required review to changes to accident analysis methods and assumptions, in particular for non-LOCA analyses. Both DG-1096 and Draft Standard Review Plan (SRP) 15.0.2 define "Evaluation Model" as follows:

- Evaluation model (EM) – Calculational framework for evaluating the behavior of the reactor system during a postulated Chapter 15 event, which includes one or more computer programs and all other information needed for use in the target application.

In discussing the evaluation model concept, the following observations are made (DG-1096, p. 3, Discussion):

The basis for analysis methods used to analyze a particular event or class of events is contained in the evaluation model concept. This concept is described in 10 CFR 50.46 for LOCA analysis but can be generalized to all analyzed events described in Chapter 15. An evaluation model (EM) is the calculational framework for evaluating

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the behavior of the reactor system during a postulated transient or design basis accident. It may include one or more computer programs, special models, and all other information necessary for application of the calculational framework to a specific event, such as:

1. ***Procedures for treating the input and output information, particularly the code input arising from the plant geometry, the assumed plant state at transient initiation,***

2. ***Specification of those portions of the analysis not included in the computer programs for which alternative approaches are used, and***

3. ***All other information necessary to specify the calculational procedure. It is the entirety of an evaluation model that ultimately determines that the results are in compliance with applicable regulations. Therefore, the entire evaluation model must be considered during the development, assessment, and review process.***

This is a broad definition. One could assume that an EM for a given accident analysis therefore consists of all of the following elements:

- **A computer code or codes like RETRAN or VIPRE or GOTHIC**, with inherent modeling, correlation, numerical solution techniques, etc. Typically, SERs are issued by the NRC on a generic basis for a given code release version, such that re-review of the internal features of the code for plant specific applications is not required. The SER typically includes statements of limitation and applicability that restrict the use of the code to a certain domain of problem types and physical conditions. It is incumbent on the user to ensure that licensing applications of the code are consistent with the qualifying statements of the SER.

  Historically, licensees have submitted topical reports for NRC review and approval that demonstrate their capability to adequately use a specific computer code or codes for safety analysis applications. More recently, the NRC issued Generic Letter 83-11, Supplement 1, which provides an acceptable process for in-house qualification of analytical models using codes which have an NRC-issued SER. Attachment 1 to GL 83-11, Supplement 1, provided elements on an acceptable in-house qualification program. These include:

  - Confirmation of prior NRC approval
  - Adequate in-house application procedures
  - A training and qualification program for users
  - Comparison or benchmarking calculations
  - Adequate quality assurance and change control processes

- **An input deck which models the plant.** These decks are typically maintained by the licensee under a configuration control process that is compatible with the quality assurance program. Modeling features may change from time to time to reflect

  - physical plant changes which have been made via either license amendment process or the provisions of 10 CFR 50.59.
  - refined or more sophisticated modeling techniques which reflect a better understanding of the physical components being modeled.
  - error corrections

- **Basic modeling assumptions.** When an input deck is modified to simulate a specific accident or transient, certain assumptions are made to ensure that the analysis is performed in a manner consistent with the design and licensing basis of the plant. Many of these accident specific modeling assumptions are specified in Section 15 of the Standard Review Plan (NUREG-0800) and reflect such considerations as the single failure criterion, appropriate treatment of the response of non-safety related equipment which could influence the event, the effects of instrument uncertainties on initial conditions and the reactor protection system and/or engineered safety features response, limiting core physics assumptions, etc.

  Additionally, many of these modeling assumptions represent accumulated years of experience in accident analyses on the part of the NRC, reactor vendors and utilities. Standard methodologies have evolved which predate the formal introduction of the Phenomenon Identification and Ranking Table (PIRT) or Evaluation Model Development and Assessment Process (EMDAP) discussed in DG-1096. The overall conservatism of the result is ensured by deterministically setting key inputs such as initial conditions, core physics characteristics, reactor protection system and engineered safety features response characteristics at the limiting ends of their tolerance bands simultaneously.

Having reviewed these basic elements of an evaluation model, it is now possible to better articulate our concerns on the scope of applicability.

1. The DG appears to propose treating non-LOCA transient analysis input decks in a manner similar to the Evaluation Model as defined in 10 CFR 50.46. This seems to be a move toward unnecessary review and oversight of licensee analysis input decks that is not needed and creates the potential for significant new overhead for both utilities and the NRC staff. In fact, it was the desire to eliminate this type of detailed model review and regulatory overhead that led to the establishment of the "50 °F rule" in 10 CFR 50.46(a)(3). As such, the Regulatory Guide should include specific allowances for a non-EM approach based on approved topical methodologies. Topical methodologies establish the licensee's ability to run codes, model phenomena, and assess results against

acceptance criteria. If a particular modeling consideration is addressed in an approved topical report, the consideration should not need to be submitted for additional review and approval by the NRC.

The NRC emphasized the value of individual utilities performing their own safety analyses in Generic Letter 83-11 and Supplement 1. However, the level of effort implied in this DG represents a daunting task that will discourage utilities from developing in-house capabilities. Cooperative development projects among the utilities through industry groups like EPRI could theoretically spread the workload, but the diversity of plant design bases and operating philosophies among the US utilities makes the development of one "generic" plant model impractical.

2. The Regulatory Guide should permit non-LOCA "base models" to be implemented as licensee evaluation models. For example, Dominion has documented and benchmarked a "base model", and we modify the variable inputs to the base model in accordance with our current licensing basis and in accordance with approved topical report methodologies. Licensees should be able to continue to justify changes to the base model from a technical and regulatory standpoint, test and benchmark the new model if necessary, and use it for reanalysis. Dominion believes that certain changes to inputs to computer codes (changes to "base models") should **not** require prior NRC review and approval prior to use and application provided the following apply:

    - **The changes reflect physical plant changes or other changes that have been evaluated** under the requirements of 10 CFR 50.59 or have been effected by license amendment request. An example would be a modification to a safety related pump head flow curve used in the model to reflect post-refurbishment test results. Suitable conservative adjustments for test instrument error are applied in developing the revised curve. An assessment of the effects of the curve change on existing accident analyses shows current results to be bounding. In this case, incorporation of the test results into the model should not require prior NRC review.

    - **The changes represent an increase in model sophistication and accuracy which has adequate technical justification**. Many simplifications are made to models for the sake of calculational convenience when the effect of the simplification is conservative or negligible. Adding points to a safety related pump head curve to better define the response in a region of interest would be an example. Another example would be to add models for passive heat sinks in the reactor coolant system which had been previously omitted on the basis that their ignoring their attenuating effects on heatup and cooldown events is conservative. The UFSAR and relevant SERs are silent on this modeling simplification. Qualification of the model changes is performed and documented consistent with the requirements of GL 83-11, Supplement 1.

As a third example, consider a case where the licensee has modeled the 5 main steam safety valves (MSSVs) on each steam line as a single lumped valve which opens at the highest lift setpoint pressure for all valves (in reality the lift setpoint varies from valve to valve). This is justified based on the conservative effect on steam pressure and RCS temperature and pressure for events which challenge the MSSVs. The UFSAR and relevant SERs are silent on this modeling simplification. The licensee now proposed to replace the lumped model with a more sophisticated one which models the correct lift setpoint for each valve. Provided adequate technical justification is made for this more realistic model, the licensee should be able to implement the change without NRC review.

- **The changes do not change the basic assumptions and considerations for ensuring a conservative analytical result as set forth in the Standard Review Plan and the UFSAR**. As an example, consider a plant that has licensed one of the new ultrasonic feedwater flow devices, such that the demonstrated calorimetric uncertainty is less than 1.0% of thermal power. Certain categories of non-LOCA transient analyses have previously assumed an initial power level of 102% of rated thermal power to reflect calorimetric power determination uncertainties. Changing the initial condition for these analysis to 101% of RTP is a model change that should not require NRC review (subject to a successful utility 50.59 review), since the basic assumption of accounting for power calorimetric uncertainty in a deterministic fashion in setting the initial conditions of the analysis has not changed.

  As an additional example, SRP Section 15.2.1-15.2.5 <u>LOSS OF EXTERNAL LOAD; TURBINE TRIP; LOSS OF CONDENSER VACUUM; CLOSURE OF MAIN STEAM ISOLATION VALVE (BWR); AND STEAM PRESSURE REGULATOR FAILURE (CLOSED)</u>, Section III, Review Procedures, specifies the following:

  > *The sequence of events from initiation until a stabilized condition is reached is reviewed to ascertain:*
  >
  > 1. *The extent to which normally operating plant instrumentation and controls are assumed to function.*
  > 2. *The extent to which plant and reactor protection systems are required to function.*
  > 3. *The credit taken for the functioning of normally operating plant systems.*
  > 4. *The operation of engineered safety systems that is required.*
  > 5. *The extent to which operator actions are required.*
  > 6. *That appropriate margin for malfunctions, such as stuck rods (see II.3.b) is accounted for.*

Consider a proposed analysis change where a PWR licensee, in analyzing the effects of a loss of external electrical load on reactor coolant system peak pressure, proposes to take credit for operation of the pressurizer power operated relief valves (PORVs), where the case currently presented in the UFSAR takes no credit for operation of the PORVs. In this case, a fundamental change in the analysis assumption is inherent in the proposed modeling change, and therefore NRC review and approval should be sought prior to using the model to change the analysis of record for this particular event. We note that this outcome would also result from proper application of 10 CFR 50.59 c(2)(viii) as revised in 1999.

The implication of the DG seems to be that virtually every model change should be subject to NRC review. This would create an unworkable situation for both the industry and the NRC and would tend to stifle model development and improvements because of the tremendous overhead that would become associated with changes.

3. The DG raises questions about scalability, similarity criteria and experimental uncertainty assessment and application to licensing methods for which there are few if any definitive answers, despite 30+ years of debate on these subjects among academicians, researchers and representatives of the NRC and industry. Reactor safety analyses are engineering evaluations which involve both art and science, judgement and analyses. No analytical approach or research database exists or will exist in the forseeable future which will remove all of these subjective aspects from the analyses.

4. In general, more clarification is needed in the Regulatory Guide of what does and does not constitutes an Evaluation Model change which is subject to NRC review and what changes to input should be controlled by the licensee under the provisions of his GL-83-11 and Quality Assurance program requirements. We note that this issue has been discussed in some detail and guidance given in Section 4.3.8 of NEI 96-07, Rev. 1 (Reference 2). Given that Regulatory Guide 1.187 (Ref. 3) has endorsed Ref. 2, it may be useful to refer to this section of NEI 96-07 in the introductory material of DG-1096 for licensee guidance on what model changes should and should not be subject to NRC review and approval prior to application to the plant's accident analysis basis.

5. If it is the intent of the NRC is <u>not</u> to apply the Evaluation Model concept to deterministic analyses, this should be explicitly stated. In other words, if the Regulatory Guide is being implemented only to address "Best-Estimate Non-LOCA Transient and Accident Analysis Methods and Uncertainty Evaluations", it should be retitled as such.

Perhaps NUREG-0800 Section 15.0.2 should be retitled "Review of Analytical Computer Codes AND MODELS", since the text indicates that the SRP section

describes "the review process and acceptance criteria for analytical models and computer codes".

If you would like further information, please contact either:

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Respectfully,

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